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Abstract

OPERATIONAL EXPERIENCE WITH PRESSURIZED-WATER SYSTEMS

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In the six years since the 1958 Geneva Conference, three power reactors of the pressurized-water type have been placed in service in the United States. These are: (1) the Yankee Atomic Electric Company reactor at Rowe, Massachusetts; (2) the Indian Point Station of Consolidated Edison at Buchanan, New York, and; (3) the Saxton Experimental reactor at Saxton Pennsylvania. Shippingport, which was already in operation in 1958, has continued steady power production since that time.

The Yankee plant first went critical in August 1960 and began power production in November of that year. Originally licensed for 392 MW(th) and 125 MW(e), three successive power increases have been authorized so that the plant is now licensed for 600 MW(th), 185 MW(e) and has been operated at that level for extended periods during the life of Core III. Two refuelings have been completed and a third is scheduled for the summer of 1964. Cores I and II were 3.4%-enriched, stainless steel clad, batch-loaded cores. Core III was a transition core, consisting of one-half re-used Core II fuel and one-half new fuel of 4.1% enrichment. Core IV will be composed of the 4.1%-enriched fuel previously used in Core III plus a half loading of new 4.1%-enriched fuel. Gross and net electrical generation in millions of kwh for the first three fuel loadings were:

	Gross	Net
Core I	1,331	1,223
Core II	1,205	1,126
Core III (est.)	1,040	970
Total	3,576	3,319

Except for down-time due to a variety of mechanical troubles in the early months of operation, and periodic shutdowns for physics testing during the life

of Core I, the plant has exhibited a high degree of reliability. For example, during 11 months of operation on Core II, the plant use factor was 88% and reactor availability better than 95%. Power and fuel costs have been more favorable than originally expected with fuel costs running in the range of 3.0 -- 4.0 mills/KWh for the first three cores.

The Indian Point Plant, which utilizes a pressurized water reactor with oil-fired superheat, first went critical in August 1962, began power operation in September and, after a six weeks shutdown to correct deficiencies in secondary plant piping, reached full licensed power of 277 MW(e) (with oil-fired superheat) in January 1963. Since that time operation has been generally reliable with some downtime attributable to condensation problems in the control-rod-drive electrical systems, mechanical problems with primary coolant pumps, primary to secondary leakage in a steam generator and the need to line the refueling cavity and canal with stainless steel to reduce water leakage. While operation with the initial core, which contains throium as the fertile material has been generally satisfactory, the results of an economic study led to a decision to purchase a second and third core using slightly enriched uranium only.

The Saxton reactor went critical in April 1962 and produced its first power in November 1962 after an extensive low-power test program. The nominal full power level of 20 MWt was attained in January 1963. Despite it being a developmental plant, more than 20 million KWh of electricity have been generated. A comprehensive test program is under way involving experimental fuel assemblies and control rods, the use of chemical shim and operation with bulk boiling in the core.

The Shippingport plant has produced well over 1-½ x 10^9 KWh of electricity since 1957. The fourth seed of the seed-and-blanket core was installed in January 1963, and operation through the life of the fourth seed will bring the accumulated average burnup of the blanket to more than 10,000 MWd/tonne. Noteworthy in Shippingport operating history, besides the fine nuclear plant performance, is the improvement made in reducing refueling time.

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Abstract

DEVELOPMENTS IN PRESSURIZED WATER REACTORS

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Since 1958, new generating capacity of 475 MWe has operated in a highly successful manner, 520 MWe is now under construction, another 1375 MWe is committed for operation before 1968, and a 1000 MWe unit will be available for 1970 operation. These units are reviewed and design features are compared. Developments in systems, components and manufacturing are reviewed in light of the effects on increase in rating and reduction in power costs. The radial peak to average power distribution generally expressed as F_T or $F_{\Delta H}$ which in the original Yankee design was set at 3.34 has been reduced through in-core measurements to determine actual operating margins and through cycled core designs as well as the demonstration and application of soluble chemical controls. For designs now underway an $F_{\Delta H}$ of 2.1 is being used with encouraging indications that additional reductions can be obtained. Modular or standardized loop and component designs are available in sizes that permit three to four fold increase in rating yet without change in the number of components. In other words, a Yankee plant could now be built in a single loop, that is, one steam generator and one pump. Reactor coolant pumps of about 65 000 gallons per minute rating and steam generators containing up to 30 000 sq. ft. of heat transfer surface are available. Reactor vessels up to about 200 inches in inside diameter suitable for cores up to about 13 feet in diameter are now considered feasible. Additional work is indicated to establish appropriate fuel management and reactivity control techniques for cores of this size. Improved fuel assembly mechanical designs tested in the Saxton experimental reactor are discussed.

System layout is being continually evaluated to reduce the reactor coolant volume and hence the stored energy which determines the design of the reactor containment system. Extensive work has been carried on to develop and demonstrate the adequacy of reinforced concrete containment systems which provide direct radiation shielding and for restricted sites double containment for stringent leakage control. The concepts of safety injection in which water is rapidly returned to the reactor vessel in the event of a loss of coolant are being developed and applied.

Progress in the design of large 1800 rpm turbine-generator designs including the application of live steam reheat between the high and low pressure turbine elements have kept the technology of the turbine electric plant in step with the size and performance requirements of the nuclear systems. The application of live steam reheat along with other cycle improvements has brought the heat rates of these machines down to about 10 000 Btu's/kwhr.

Experience with, and designs of, system safety features and engineered safeguards are examined. Siting requirements and designs to meet them are reviewed. Possibilities for further improvements, notably cost reduction without sacrifice of safety, are discussed.

The author reviews typical AEC and industry sponsored programs and illustrates the ease with which such developments as "chemical shim" (the use of neutron absorber dissolved in the reactor coolant) have been incorporated into existing as well as new designs. Predictions of the future of PWR are made, including the possibilities of using super critical rates.

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Abstract

OPERATING EXPERIENCE WITH BOILING WATER REACTORS

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The operating experience of three utility-owned and -operated reactors will be included in this report: Commonwealth Edison Company's Dresden, Pacific Gas and Electric Company's Humboldt Bay, and Consumers Power Company's Big Rock Point.

These reactors represent different design concepts. Dresden is a large dual-cycle boiling water reactor, designed for maximum flexibility of operation on a large electric utility system. It combines the features of both pressurized and boiling water reactor types. The Big Rock Point reactor is a forced-circulation boiling water unit designed for high-power-density fuel and operation at variable pressures in order to advance reactor technology. The Humboldt Bay unit was designed to produce economic nuclear power in its area. This has been accomplished by elimination of heat exchangers and recirculating pumps in the design of the natural-circulation boiling water reactor.

Nuclear performance tests have verified the inherent safety features of the boiling water reactor. The large boiling water reactors have demonstrated that cores having many critical masses are stable and easy to control. The control-rod arrangements are such that the worth of individual rods is minimized and power distribution easily controlled. Very little rod motion is required to compensate for xenon variations, even in the most severe cases, and there has been no instability due to xenon.

The reactor water chemistry of boiling water reactors has been found to be relatively simple. No chemicals need to be added to the reactor water system. The main criterion for reactor water treatment is complete demineralization.

Radiation levels with the plants at full power have demonstrated the adequacy of the shielding provided for both direct and scattered neutrons and

gammas. Discharge of radioactivity to the environs has been closely monitored at all plants. The levels of activity discharged in all cases have been remarkably low -- well below the limits specified by the United States Atomic Energy Commission.

Fuel-handling experience has indicated that simple fuel-handling mechanisms, together with good plant layout, can minimize outage time. Experience gained at Dresden has resulted in improvements on the newer reactors, such as eliminating the need for moving fuel in and out of the containment vessel during shutdown, and by providing bottom-entry in-core instrumentation.

Early mechanical difficulties encountered at both Dresden and Big Rock Point were solved satisfactorily. At Dresden the control-rod drives had to be modified early in operation. Inspections conducted after a substantial period of power operation revealed that the original problems had been solved; the improved designs were incorporated in later reactors. At Big Rock Point excessive vibration of the core fuel-support tube and channel assemblies was detected during early testing. It was necessary to unload the fuel to facilitate modifications to correct the difficulty. Subsequent tests revealed that these modifications produced satisfactory performance.

Maintenance of these plants during operation has consisted primarily of reparing steam or water leaks on conventional valves and equipment. Control-rod-drive maintenance has posed no major problem. The improved control-rod drives on the Big Rock Point and Humboldt reactors can be removed, serviced and replaced without removing the vessel head. Turbine overhaul after extended reactor operation, including operation with known leaking fuel assemblies, is not significantly different from that of a conventional plant.

The ability of these plants to follow utility system load schedules has been definitely established. The actual operation has proved to be relatively simple, and automatic load control is planned for at least one of these units in the near future.

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Abstract

DEVELOPMENTS IN BOILING WATER REACTORS

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Since the second Geneva Conference in 1958 on the Peaceful Uses of Atomic Energy, boiling water reactor power plants have been accepted as a commercial reality competing with other forms of energy for electrical power production in several countries. This paper discusses the key factors that have brought about the evolution of this reactor concept.

Today, 12 U.S.-designed boiling water nuclear power stations have been commissioned and operated, representing 584 300 installed net kw. Six new BWR nuclear power stations, adding 1 995 000 net kw of installed capacity, are in various stages of design and construction. Initial designs were conservative as nearly every reactor placed into operation is capable of producing more power than initially expected.

The evolution of the boiling water reactor has followed a process of simplification in design resulting from operating and test performance. Key technical features which are important to equipment simplification and improved safety include the use of the single cycle with steam separation inside the reactor vessel and a pressure suppression type of containment. Much emphasis is placed on reactor control for safety and on the reliability of nuclear equipment.

Observed results from the operation of BWR cores has demonstrated the practicality of achieving a considerable degree of power flattening by proper control rod programming and scatter loading of replacement fuel. The data accumulated allows accurate prediction of lattice parameters and reactivity performance. Earlier segmented fuel rod designs are superseded by straight-through rods. Currently, reactors are licensed with burnout margins of 1.5 in contrast to earlier margins of 2.0. Design core exit steam qualities have increased from 5 to 9-1/2 weight per cent.

In the BWR, Zircaloy is the most suitable material for fuel element cladding from the standpoint of overall performance. The direction of development is toward thinner cladding for economy with integrity to burnup exposures in the range of 20 000 MWD/ton. Experimentally, fuels have demonstrated satisfactory performance above this range.

Some early fears concerning the boiling water reactor are proven to be unfounded. The use of reactor steam directly in the turbine has been satisfactory without excessive radiation, corrosion or maintenance problems. The reactor has operated stably with high voids and responded well to all transient operating conditions. Core power distributions are readily measured and regulated. The handling of off-gases and other waste is satisfactory - well within established legal limits.

Future trends in the BWR are in the direction of further equipment simplification, higher core power densities and improved steam conditions. Reduced movable control requirements, improved neutron economy, flatter power distributions and greater exit voids are the major directions for improvements of BWR core designs. The use of nuclear superheat is being vigorously explored with four nuclear superheat prototypes in construction or in operation. Each reload of reactor fuel incorporates the latest technology and economic design changes. Progressive automation of nuclear plant operation can be expected. The trend toward larger unit sizes has been rapid and will continue.

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SODIUM-COOLED THERMAL REACTORS

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The objectives of the development program for sodium-cooled thermal reactors are 1) attainment of low-cost electric power; 2) a reactor type having exceptional safety characteristics; 3) development of large sodium systems for future fast breeders. The U. S. Atomic Energy Commission program includes the operation and power expansion of the 20 MW(th)/6MW(e) Sodium Reactor Experiment since 1957, the 256 MWth/81 MW(e) Hallam Nuclear Power Facility on the system of Consumers Public Power District of Nebraska, and the development of major components, uranium monocarbide fuel and plant designs for 200 and 1000 MW(e) Sodium Graphite Reactor power stations.

Current designs utilize enriched UC fuel and produce steam at 170 atmospheres and 570C. Because of the low reactor pressure and the chemical affinity of sodium for iodine, there is little fission-product release from the system in event of fuel failure, thereby permitting conventional buildings. The high thermal efficiency, near 43%, results in good long-term fissile fuel utilization. A 500-MW(e) plant of this type can produce power, under USA private financing conditions, at a projected 5.7 mills/kWh.

The SRE began operation in 1957. The first core was unalloyed uranium metal, and operated at sodium temperatures up to 1000F(538C). In 1959, twelve fuel elements failed due to coolant flow blockage by in-leakage and decomposition of auxiliary hydrocarbon coolant. The damaged core elements were removed with remote-handling equipment, and operations resumed in late 1960, with a second core of metal Th/U alloy and the original Na coolant. Subsequent operations have provided extensive information on contaminant removal from Na systems, dynamic behavior of various fuel elements, and on performance of components in sodium.

The Hallam Nuclear Power Facility began power operation in July 1963. The first core is U/10% Mo alloy; the second fuel loading being fabricated is UC. With Na at 930F (500C) steam is produced at 60 atmospheres, 815F (435C). During preoperational tests, a tube failure occurred in an intermediate heat exchanger due to flow-induced vibration. All six exchangers were satisfactorily modified with internal bracing. The Zircaloy control-rod thimbles had to be replaced with stainless steel due to internal hydriding. All other major components and systems have performed satisfactorily.

The plant has operated up to 81 MW(e) under load-following conditions on a routine basis since October 1963. By January 1, 1964, about 80 million kWh of electric energy had been produced. The plant availability during the previous month of December 1963 was 99.4%.

Large SGR designs are now being developed in the 200 to 1000 MW(e) range. The fuel is stainless steel clad UC operating at a power density of about 50 kW/kg. With sodium coolant temperature at 1500F (620C), steam pressures can be between 170 atmospheres and supercritical. The graphite moderator for these plants is now contained in a single large stainless steel calandria vessel. The heat-transfer systems are designed for about 500 MW(th) each, using single large 2200 liter/sec pumps, and modular once-through steam generators. A 200-MW(e) plant uses one such loop, a 1000-MW(e) plant five similar loops. Advanced variants of the SGR, e.g., coupled thermal-fast cores, may provide near-term intermediate steps to fast reactors.

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FAST POWER REACTORS

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The development of Fast Power Reactors continues to be motivated by their capability to breed and thus achieve efficient utilization of nuclear resources. This capability has been placed in better perspective since the 1958 Geneva Conference as a result of estimates which have been made of future power requirements in the United States and the role of nuclear power in meeting these requirements. These estimates indicate that Fast Reactors will be needed in the 1980's. The U. S. program is directed toward the development of Fast Reactors which will be compatible with electric generating system requirements at that time.

It has been estimated that the present trend toward larger power stations will continue and that in about 20 years stations of approximately 1000 MWe size will be desired. The U. S. program has been altered to emphasize the development of these large plants. Four independent design studies of 1000 MWe Fast Reactor Power Plants were recently completed for the U. S. Atomic Energy Commission. This paper reviews these studies with particular emphasis on the indicated trends in reactor design and the indicated future development requirements.

The major fast reactor facilities now available in the U. S. are the EBR-II and Enrico Fermi Atomic Power Plant. Construction of these facilities has been completed and initial operation has commenced. Difficulties were encountered which delayed their operation but indirectly this experience contributed significantly to the technology. In particular, valuable maintenance experience has been obtained with the sodium systems. At EBR-II, the primary sodium pumps, a control drive, the fuel gripper, and other

mechanisms have been removed from the sodium system. At Fermi, pumps, steam generator internals, and the offset handling mechanism in the fuel handling system also have been removed. All of these operations were performed without violating the inert gas atmosphere or significant contamination of the sodium. Special procedures were developed for these operations which will be described. In addition, major repairs and modifications were made inside the Fermi reactor vessel after draining the sodium. This was accomplished with a minimum of contamination of the inert gas environment by men working in air-tight suits with external air supplies.

The experience with sodium filling and sodium system operation has been extremely encouraging. The 80 000 gallon EBR-II primary system was filled without incident and sodium quality has been maintained without difficulty. The same filling experience was obtained at Fermi and while some contamination with oxygen, hydrogen, and carbon occurred during preoperational testing, it was removed without difficulty. At no time was difficulty experienced with plugging of systems in either facility.

All of this experience was obtained on large components and systems and is applicable to power station operation. It is particularly relevant to the plans for larger plants involving larger capital investments.

The initial performance of both EBR-II and Fermi have been very encouraging. There has been excellent correlation between predicted performance and actual results. This applies equally to those parameters that were checked by critical experiments such as critical mass and control rod worth as well as those dependent primarily on analysis such as temperature coefficients and sodium coefficients. The nuclear characteristics were found to be as predicted and indicate that acceptable power performance can be expected.

Power operation of these facilities will provide additional data and experience, including power induced reactivity effects, irradiation data and over-all plant operational experience. Current status and operating data as well as future programs for these facilities are described.

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Abstract

SEED AND BLANKET REACTORS (EXPERIENCE, DISTINGUISHING FEATURES, PHYSICS, LONG RANGE POTENTIAL)

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R. G. Luce (Knolls Atomic Power Laboratory)

The basic features and advantages of seed blanket reactors will be presented. The concentration of highly enriched fuel in nearly critical seed regions results in reduction of enriched fuel requirements, centralization of control, and the feasibility of tailoring the nuclear environments to the optimum both for the highly enriched and for the fertile fuels.

The Shippingport Reactor Core 1 recently completed operating with its fourth and last seed. Except for minor replacements and relocations performed during refueling of the seed, the natural uranium in the blanket region of the core has remained in place throughout the entire lifetime of the core. At the time of each seed refueling, several blanket fuel elements have been removed from the core for study. Isotopic analysis of the fuel material in these rods provides a basis for evaluation of the techniques employed in calculation of the behavior of natural uranium during extended irradiation in a seed-blanket arrangement.

The most straightforward and most meaningful method of employing the experimental data to evaluate a design depletion calculation is to compare the experimental results for each individual rod directly with the isotopic concentrations given for each rod by the calculation.

The results indicate general adequacy of the calculational model for the U-235 depletion and build-up of plutonium isotopes. Although there are isolated discrepancies of as much as 20%, the average difference between calculation and experiment is well within 10%.

The advantages of the seed-blanket core concept are enhanced in large reactors. For example, the core can be controlled by varying the seed geometry, without the use of parasitic poisons. Still further gains can be obtained by application of the seed-blanket concept to the thorium cycle. Several thorium cycle designs are discussed: a thorium converter burner core designed for ten years' operation without refueling; a converter recycle design optimized for breeding; a thorium burner core with the central region optimized for breeding.

The effects of long recycle of the U-233 fuel to equilibrium composition is discussed, including evidence that the seed-blanket breeder core design can continue to breed even after over 100 years of fuel recycling.

The experimental work and theoretical methods which have been developed for seed-blanket cores are described and discussed.

The paper concludes with a brief comparison of the potential of the seed-blanket core with other types which have been proposed for large nuclear power plant operation.

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Abstract

HEAVY-WATER POWER REACTORS AND ORGANIC-COOLED POWER REACTORS

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Nuclear power plants employing reactors moderated with heavy water and using a number of different coolants have been investigated extensively in the United States in the past five years. The power plants studied are of progressively larger outputs, the majority being of 500 MW(e) size. The reactors use pressure tubes and employ graded irradiation of the fuel in order to limit excess reactivity. On-power fueling also is a feature of many of the designs. The coolants are liquid D_2O, boiling D_2O, fog H_2O, pressurized CO_2, and organic fluid. In the present paper, estimated capital, operating, and fuel-cycle costs are given for each type of coolant and for several financing schemes. Fuel-cycle costs in the range of 1 mill per kilowatt hour or lower are obtained for most coolants with total power costs in the range of 5 to 6 mills/kwh using utility-company financing. Lowest costs are obtained when the fuel is slightly enriched uranium at U. S. prices. Energy costs for natural-uranium fuel also are shown. The effect of chemical processing charges and Pu credit on the power cost is reviewed, and the cost advantage or penalty for a throw-away fuel cycle is shown.

Fuel-element designs include cylindrical UO_2 pellets clad in metallic jackets, massive annular cylinders fabricated by vibratory compaction of UO_2, and metallic uranium clad with bonded jackets. Results of studies are given that explore the potential of reactors fueled with thorium and U-233 and moderated with D_2O for operation as breeders.

Organic cooling has received special attention since the studies show that heavy-water moderation with organic cooling will yield both low capital and fuel-cycle costs. Experiments have shown that the formation of deposits (fouling) on fuel-element surfaces has been controlled by maintaing coolant purity below 5 parts/million ash by means of distillation, filtration, and

adsorption. The feasibility of fabricating fuel elements clad with sintered aluminum power (SAP or APM) has been demonstrated. Operation of the Organic Moderated Reactor Experiment, in which 98 SAP-clad UO_2 prototype fuel rods were irradiated to 11 000 MWd/tonne at cladding surface temperature up to 850F (450C) without any fuel failures or deposition of fouling films gives concrete evidence of the effectiveness of coolant purity control and the integrity of the SAP-clad fuel. Operation of the Piqua, Ohio Nuclear Power Plant since November 1963 has confirmed the feasibility of reactor cooling by organic fluid in systems using carbon steel as a material of construction. Studies of the D_2O-moderated organic-cooled system have indicated advantages for the uranium carbide fuel. Out-of-pile tests have shown the compatibility of properly-treated SAP with uranium carbide fuel. The feasibility of zirconium alloys for organic service still requires definite confirmation. Alloys and protective surface layers are promising.

The role of the heavy water reactor in conserving fissile material resources is reviewed, including its potential for Pu recycle. Also, the effect of the low inventory of fissile material of the heavy-water reactors on uranium utilization is illustrated, in particular where the heavy-water reactor is combined with breeder systems.

Since there appears to be no size limitations to heavy-water-moderated reactors, estimates of energy costs for power plants of 1 000 MW(e) and larger have been made and show a continuing trend to lower capital and fuel costs.

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DEVELOPMENTS IN SPECTRAL-SHIFT REACTORS

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The spectral shift control reactor is a variation of the pressurized water reactor which shows promise of reducing both fuel and capital costs significantly. In this reactor a large portion of the excess reactivity which must be built into a PWR core to achieve long core life is balanced out by modifying the neutron spectrum so that resonance absorption in fertile material is markedly increased. This "shift" of the neutron spectrum is produced by mixing a certain amount of heavy water with the ordinary water in the primary system. As the core is operated the concentration of heavy water is decreased to reduce resonance absorption and thus make more neutrons available to continue the chain reaction.

By shifting the neutron spectrum towards the resonance region at the beginning of operation with a new core, the fertile material acts as a control rod except that the neutrons which would normally be lost to the fuel cycle produce new fuel. This results in a significant improvement of the conversion ratio - in particular when operated with the Uranium 2-33/Thorium fuel cycle near breeding.

The use of a uniform method of shim control eliminates the distortions of the power distribution caused by control rods, making it possible to obtain a high degree of power flattening over the entire core life by proper distribution of fuel in the reactor core. The resulting improvement in power density results in significant savings in plant capital costs.

The principle of spectral-shift control has been confirmed by an extensive series of critical and exponential experiments. Data from these experiments, which have been in progress since July 1960, are presented and include measurements of buckling, cadmium ratios for fission and capture in fertile material, thermal-disadvantage factors, reflector savings, neutron age, change of buckling with temperature and neutron spectra. These

measurements have been made in several lattices at several concentrations of heavy water for 2½%-and 4%-enriched UO_2 and 93%-enriched UO_2/ThO_2 fuels.

The experimental data are correlated by a multigroup diffusion model which uses a Greuling-Geortzel slowing-down kernel. Resonance absorption is calculated by methods similar to those developed by Chernick and Nordheim and is normalized to agree with measured resonance integrals.

Design optimization studies are presented, with particular emphasis on minimizing generating cost. Various fuel-management schemes are considered. The effect of spectral-shift control on the conversion ratio and on uranium utilization is presented.

The economic potential for spectral-shift reactors operating on the uranium-233/thorium fuel cycle is compared with ordinary water reactors. The main features of a prototype plant, the construction of which is expected to begin in 1964, and plans for demonstrating the uranium-233/thorium fuel cycle in the prototype plant are presented.

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SMALL POWER REACTORS AND PORTABLE REACTOR POWER PLANTS

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The United States program in Small Power Reactors and Portable Reactor Power Plants, other than for space applications, is concentrated generally in the Army Reactors Program--a joint program of the Atomic Energy Commission and the Department of the Army.

The low power levels appropriate to the requirements (1-10 MWe) of this program plus the requirement for system portability introduce design limitations. The portability requirement means that the plant must be of minimum size and weight and be packaged in modules which can be carried in available transport aircraft. The possible utilization of these plants at remote sites of minimal accessibility requires that plant modules be preassembled and tested at the factory to reduce site installation effort and that plant installation not require extensive heavy construction equipment. This type of application also places heavy emphasis upon high plant availability and reliability of operation. Simplicity of plant operation is emphasized also by the necessity that the plant be operated by a small crew not possessing a high degree of specialized training.

Design characteristics typical of these plants are illustrated by the PM-1 located at Sundance, Wyoming, USA and the PM-3A at McMurdo Station, Antarctica. These totally self-sufficient plants producing 1-1.5 MWe can be transported in 18 air transportable packages of plant equipment and about 40 packages of support facilities and buildings. The PM-3A was fabricated and preassembled at the factory in 16 months, including operational test of the plant on non-nuclear heat. It was installed in a previously prepared site and taken critical 77 days after arrival with a total installation effort of 5 447 man-days.

Following test operation, both of these plants have characteristically been at power about 85% of the time adjusted for scheduled shutdowns. They can be routinely operated by a crew numbering about 16 men. Fuel cycle costs are estimated to be in the range of 20-35 mils/kwhr.

The design technology and operational experience being developed in this program is applicable to many non-military installations.

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Abstract

THE U. S. PROGRAM FOR THE DEVELOPMENT OF ADVANCED CONCEPTS

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The United States Atomic Energy Commission is currently investigating a number of reactor concepts that offer significant long-range promise for advancing reactor technology. Each has potential utilization for one or more applications -- civilian central station plants, space power units, maritime propulsion units, military package power plants or others. To start the early exploration of these attractive areas of technology without becoming committed to the costly programs associated with urgent requirements and schedules, the Commission has undertaken the deliberate development, design, construction and operation of a number of relatively simple reactor experiments whose primary purpose is to demonstrate and establish the basic technical feasibility of the concepts involved and to provide an adequate basis for realistic evaluation of their ultimate potential. At present, the concepts under investigation include the Molten Salt Reactor Experiment, the Experimental Beryllium Oxide Reactor, the Ultra High Temperature Reactor Experiment, the Los Alamos Molten Plutonium Reactor Program, the Medium Power Reactor Experiment, and the Advanced High Temperature Gas Reactor, as well as others. A portion of the program is directed toward investigations of direct power conversion methods.

The Molten Salt Reactor Experiment (MSRE) is a 10 thermal megawatt experiment at Oak Ridge National Laboratory. It uses a mixture of molten fluoride salts as a combination fuel-coolant with graphite as the moderator. For thermal- or fast-neutron systems, this concept offers the potential of an economic, high efficiency system with high temperatures and specific power and low fuel cycle costs. The MSRE is scheduled to commence nuclear operation in 1965.

The Experimental Beryllium Oxide Reactor (EBOR) is being developed by the General Atomic Division of General Dynamics and is now under construction at the National Reactor Testing Station in Idaho. This 10-MW(th) experiment will explore and provide detailed data on the use of beryllium oxide as a high temperature reactor moderating material for long-term application. The EBOR is also scheduled for 1965 operation.

The Ultra High Temperature Reactor Experiment (UHTREX) at the Los Alamos Scientific Laboratory is a 3-MW(th) project to demonstrate the operation of a gas-cooled reactor up to temperatures of 2200-2400F (1220-1320C). This experiment is helium cooled and graphite moderated and will operate on unclad fuel to develop basic data that could be applicable to process heat or magnetohydrodynamic systems. The experiment is planned for 1965 operation.

The Los Alamos Molten Plutonium Program is generating basic data to evaluate the use of molten plutonium fuel in reactors. The Los Alamos Molten Plutonium Reactor Experiment (LAMPRE), 1-MW(th) experiment, has been in operation since 1961. A larger facility, the Fast Core Test Facility (FCTF), started construction in 1963, is a 20-MW(th) with provisions to accommodate a separate blanket region. The FCTF will be a flexible core test experiment and is scheduled to commence operation in 1967.

Development work is under way on the Medium Power Reactor Experiment, a potassium cooled, stainless steel reactor concept, whose objective is to demonstrate advanced technology applicable to a space power system of 100 to 300 kW(e) with specific weights on the order of 30 lb/kW(e) using conventional materials and reactor temperatures of about 1500F. Research and development is also being conducted on the Advanced High Temperature Gas Reactor, a fast-spectrum refractory-metal concept with potential for space power and other compact power system applications.

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Abstract

HIGH-TEMPERATURE GAS-COOLED REACTORS

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The first HTGR 40-MW(e) power plant in the U. S. is nearing completion at Peach Bottom, Pennsylvania. Design and experimental work on full-scale HTGR plants ranging from 200 MW(e) to 1000 MW(e) have been carried out since 1960. Research and development for Peach Bottom have been intensively conducted for the past 5 years under a joint program between the U. S. Atomic Energy Commission and General Atomic. The Peach Bottom 115-MW(th) reactor employs all-graphite, vented fuel elements with coated U/Th carbide particles and with surface temperature up to 1100C. Helium from the reactor at 750C generates superheated steam at 538C by means of a new design of nuclear steam generator.

Other new features in the construction of Peach Bottom include (a) appropriate helium-coolant circulation equipment; (b) fission-product trapping and helium-purification system; (c) new type of refueling equipment; and (d) control rods of B_4C dispersed in graphite for high-temperature operation with hydraulic motor drives.

Research and development for the HTGR have involved all phases of reactor physics, materials development and component design and testing. Results of work in the areas of neutron thermalization in crystalline moderators and in resonance capture theory have been incorporated in an extensive library of reactor-core computer codes. Critical-facility experiments were conducted and checked closely with calculation. These were used to obtain design data for the new semi-homogeneous, graphite core design and for verifying temperature coefficients and control rod worth. Materials development work has included extensive testing, both in-pile and out-of-pile, of low permeability graphites, graphite matrix fuels, and pyrolytic-carbon-coated fuel particles of (U,Th) dicarbide. Results on more than 40 fuel compacts and several hundred samples

of graphite irradiated in capsules under different conditions of exposure and temperature have been obtained.

Proof testing of full-diameter graphite, vented fuel, as well as the associated fission-product-trapping and helium-purification principles, has been carried out for 35 months in the GAIL loop. As of January 1964, fuel elements have reached 37 500 MWd/ton average and 50 000 MWd/ton peak burn-up.

Prototype fuel production experience on a pilot line has included fabrication of almost 2000 kg of coated fuel particles during the latter half of 1963 for the first Peach Bottom core. Good results were obtained in meeting specifications for particle structure, composition and integrity.

The Peach Bottom plant is being built for Philadelphia Electric Company and 52 utilities of the High Temperature Reactor Development Associates group. The entire nuclear steam supply including the reactor and fuel elements are being furnished by General Atomic. Bechtel has engineered the power plant and is constructing it.

General Atomic design and development for larger HTGR plants have included work on 1) a 500-MW(e) plant scaled up from the prototype plant employing a six-steam-generator-loop system and new compact arrangements in the reactor core/vessel and in the over-all plant layout; 2)work on a 250-MW(e) plant of more advanced design employing a concrete pressure vessel enclosing the entire primary circuits; and 3) work on a 1000-MW(e) reactor with concrete pressure vessel with special attention to the possibility of thermal breeding in the U-233/thorium cycle. These programs are being undertaken respectively with the ESADA group of utilities, the ARDA group of utilities, and the USAEC.

There has been extensive development and testing of the graphite fuel to the equivalent of six-year in-core life and fabrication and testing of control drives and fuel-handling equipment for the large plants. Detailed large HTGR designs have efficiencies of 40 to 45 percent using conventional plant steam cycles with reheat. Larger size also improves neutron economy and, with improvements in fuel cycles, results in conversion ratios in the 0.8 to 1.05 range. Projected fuel-cycle costs for 500 MW(e) plants are in the range of 1 to 1.5 mills per kilowatt hour.

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Abstract

THORIUM UTILIZATION SYSTEMS

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The use of thorium in nuclear power reactors is of interest because the uranium-233 bred into the thorium has a better neutron yield in thermal power reactors than either uranium-235 or plutonium-239. Fuel systems based on thorium and recycled uranium-233 might therefore be expected to have higher conversion ratios and longer fuel lifetimes than natural uranium or slightly enriched uranium fuels. These attributes, in turn, might possibly result in lower power costs. In certain reactor types moreover, it may be possible not only to achieve low power costs but also to breed uranium-233 sufficiently to achieve a self-sustaining recycle system whereby no supplementary input of fissile material would be required.

Because of these possible advantages, thorium-fueled reactors might constitute an important segment of future nuclear-power systems; however, the integration of such reactors into any existing nuclear industry requires that they be economically competitive with reactors based on the use of natural uranium or slightly enriched uranium. It is important, therefore, that thorium-utilization systems have the combined capability of being economically competitive for the short term and of making efficient use of thorium for the long term. These two criteria form the basis for the evaluation and selection of reactor types that might be most promising for developing a nuclear industry involving the use of thorium. This paper summarizes the design characteristics of various thorium converter reactors and reviews the results of recent studies and evaluations of their economic potential. Thermal converter types considered include light-water seed/blanket, spectral-shift controlled, heavy-water moderated, high-temperature gas cooled, and graphite-moderated molten

salt. Data are given to indicate the nuclear performance and associated potential of each concept for achieving competitive power costs during the period 1970-1980 and at the same time conserving supplies of low-cost uranium. The potential of each system for achieving self-sustaining fuel recycle and utilization of high-cost ores is also indicated. Factors that influence power costs in these high-neutron-economy converters such as, conversion ratio, fuel specific power, recycle fuel composition, fuel processing rates, method of processing, unit fuel cycle costs and economic ground rules are also discussed.

To achieve the long-term goal of developing an economic nuclear industry based on the use of high-cost ores may require thorium breeders with even better nuclear performance and lower unit fuel processing costs than the high-neutron-economy converters described above. Very little development work aimed specifically at such thermal breeders is at present under way in the United States; however, a number of concepts have been designed and evaluated for the purpose of establishing the incentives for initiating development programs. Concepts considered include aqueous homogeneous slurry breeder, two-region molten-salt breeder, liquid-bismuth breeder, gas-cooled graphite breeder, and gas-cooled heavy-water breeder. Data on fuel cycle costs as a function of breeding ratio, fuel yield (or inventory doubling time), and fuel specific power are presented to indicate the breeding potential of these concepts and their outlook for achieving economic power from high-cost nuclear source materials.

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Abstract

DEVELOPMENTS IN NUCLEAR SUPERHEAT

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As is well known, the trend towards higher steam temperatures during the past half-century has resulted in an improvement in efficiency and decrease in power production costs in conventional turbine-driven electrical generating plants. By increasing the temperature of steam from boiling water reactors through nuclear means and utilizing the advanced technology in materials and equipment, it should be possible to likewise lower the cost of converting nuclear to electrical energy.

In order to investigate and solve the problems involved in superheating steam by nuclear means, the U. S. Atomic Energy Commission instituted an engineering and development program in 1958 in this matter. From the results of studies and the construction and operation of demonstration reactors, the most feasible and economic method of accomplishing this was expected to result.

It was anticipated that the feasibility and economics of superheating steam would depend on the problems encountered and methods selected to solve them. It was clear that certain factors might result in increased cost which, in turn, would have to be balanced against the advantages to be gained.

The problems which face the investigators in this program are discussed and the results of work which have been accomplished to solve them are presented in this paper. One of the anticipated problems is related to reactor safety and concerns the expected large change in reactivity in the event the normally voided superheated steam passages in the fuel elements are flooded with moderator water. The methods used to reduce the magnitude of the change or to reduce its rate are presented. Another factor with which the reactor designer must be concerned is that of the partition of power between the boiling and superheating regions of the integral reactor. Since burn-up of fuel and

fission product build-up are expected to change this division of power with time, methods of controlling and adjusting it must be incorporated and are discussed.

Since the desired steam temperature and reactor environment will result in fuel element cladding temperatures and conditions which present a materials performance problem, this matter is also considered and discussed.

In the normal operation of a boiling water reactor, reductions by factors as high as 10^4 in the content of radioactive contamination in the steam compared to that in the reactor water occur. This advantage is not expected to pertain in the superheater region when the outlet steam is compared to the inlet steam. Thus, any release of fission products or mass transfer of fuel cladding material would be expected to pass quantitatively to the energy conversion system and present operational and maintenance problems for such items as turbines, condensers, demineralizers, and their associated piping systems. This matter is duscussed in this paper, as is the development of water-steam separation devices at the superheater inlet.

Various studies of superheating reactor designs were undertaken in this program; and, from them, a number were selected for construction. These include designs which incorporate integral regions where steam generated in the boiling region is passed directly back through the superheating region in either the center or periphery of the core, as well as the design of a separate superheating reactor. Designation of them is given as EVESR, BONUS, BORAX V, and Pathfinder; and substantial operating experience with them will have been received by mid-summer 1964. Except for EVESR, these reactors are part of completely integrated electrical generating plants and, therefore, also serve to demonstrate the feasibility of operating such plants with superheated steam. Pertinent descriptions of each reactor, as well as the results of operational experience to date, are given in this paper.

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Abstract

INTEGRATION OF NUCLEAR REACTORS AND POWER NETWORKS IN THE UNITED STATES; ECONOMIC AND TECHNICAL ASPECTS

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Primary considerations in the selection of an energy source for an electric generating station are: (1) cost - of competitive heat and power supplies; (2) dependability - that electricity will always be ready; (3) availability - in any quantity to anyone who may desire it at any location; (4) economy - in operation and use; and (5) quality - to meet the existing voltage and frequency regulation demands of modern equipment and controls.

Freedom in selecting primary energy sources is increased by the existence of large regional electrical networks which interchange power on an economy as well as an emergency basis. Network operation (81% of this country's generating facilities are tied together in one interconnection and the remainder are in a second network) helps smooth the peaks and valleys of demand but also places a more insistent call on individual generating stations to follow closely power swings and surges.

In view of the high capital investment needed for a modern electric generating station, the question of timing becomes very important. It is desirable to bring a new station on stream at the exact time when reserve capacity reaches the minimum compatible with supplying quality service. Optimum reserve margins vary between 10 and 15% of total system capacity and are often established by statistical techniques rather than fixed percentages.

Energy sources for base load stations, in addition to nuclear, are primarily fuel oil and coal in the coastal states; oil and gas in the southwest; hydro in the northwest and coal in the remaining areas. The size and location of the market for individual fuels has been changing rapidly in the past two years due to new concepts in transportation of

the fuel, such as integral trains, pipelines, conveyors and turn-around barges; and in transmission of power at voltages of 500 KV and above.

The optimum location of a base station is the point where line losses can be minimized and voltage maintained at a constant standard. This point must be correlated with land availability, safety and air pollution. After balancing all of the above factors, the ultimate decision becomes one of the cost of a kilowatt at the customer's meter. Generally, this cost is divided into fixed and operating costs, with a very small percentage going to administrative and general management expense.

Fixed costs include interest on funds borrowed for construction, depreciation, taxes and insurance and vary from 10 to 14% of the plant investment annually over the amortized life (30-50 years) for investor-owned utilities. Fixed costs for government plants vary from 5 to 7% annually. This difference in fixed costs tends to make development of any power facilities superficially more attractive to the government sector than to the private sector as the former pays less interest on borrowed money and, of course, is exempt from taxes, thus shifting the power cost in part to the taxpayer rather than the consumer.

Operating costs are sub-divided as follows: operating labor and supplies; maintenance, labor and materials; and fuel costs, including purchase price, transportation and handling charges. By relating fuel costs to heat value, consideration is given to developments in fuel processing and handling. For nuclear, this involves private ownership of fuel and gains in breeder technology. For fossil fuel it includes lowering of transportation costs as well as improvements in technology, such as higher temperatures and pressures, which will increase boiler efficiency.

Although difficult to evaluate, final consideration must be given to future availability of the primary energy source, that is, whether coal, oil, uranium and/or plutonium be available in adequate quantities at satisfactory cost levels until amortization of the specific plant has been completed.

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ISOTOPIC POWER SOURCES COUPLED WITH THERMOELECTRIC CONVERTERS

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Thermoelectric generators fueled with radionuclides are supplying power dependably today in space and remote terrestrial systems. This technology, developed under the aegis of the USAEC's SNAP Program (Systems for Nuclear Auxiliary Power), was shown to be feasible in 1959. Early operational status was achieved in 1961 when it was selected to provide power to a satellite and an unattended meteorological station 700 miles from the North Pole. Each system has logged more than 2 years of continuous trouble-free operation.

Power in these systems is obtained through absorption of the decay energy of the selected radioisotope and its degradation into heat. The thermal energy is then conveyed through a heat engine, such as a thermoelectric converter, producing electricity. Unconverted heat is rejected from the generator system, in a manner suited to the application of the device, e.g., radiation in space, and conduction and convection in terrestrial missions. In suitable combination, the technology yields a power supply that is extremely compact, reliable and light-weight, for lifetimes of 10 or more years in the power range of less than 1 watt to a few kilowatts. Space systems have already achieved power-to-weight ratios of greater than 1 watt per pound with the potential of doubling this figure in the next few years.

Design of the power system requires consideration of the application (space or terrestrial) and its lifetime, safety analyses, selection of the fuel (alpha- or beta-gamma-emitters), thermoelectric materials, insulation, shielding, heat-rejection mechanisms and power conditioning.

Useful radionuclide fuels are obtained from two sources: separated and purified fission products, or through neutron irradiation of target materials (e.g., curium-242 from irradiation of americium-241). Selection criteria include half-life, power density (thermal watts per cubic centimeter), physical and chemical form, cost and availability. Radioisotopes in the quantities required for power generation would create a hazard of some significance if improperly released to the biosphere. Safety, therefore, is a prime factor in establishing the practicability of this technology and assurance must be established for each application that the use of a power source will be harmless to man. Finally, a basic characteristic of a thermo-electric converter is that it is a high-current, low-voltage system requiring series connection of the thermo-electric elements to increase the overall voltage. A power conditioner is required to match the electrical out put of the generator to that of the payload. Static solid state power conditioners have been built operating at efficiencies up to 85% between 3 and 30 volts. These radionuclide fuels include Sr-90, Cs-137, Ce-144, Pu-238, Cm-242, Po-210 and Cm-244.

The United States has several of these power sources in various stages of development. They include Transit 4-A and -B (2.7 Watt, Pu-238), SNAP 9A (25 Watt, Pu 238), SNAP 11 (25 Watt, Cm-242), SNAP 13 (12 Watt, Cm-242), Weather station (4.5 Watt, Sr-90), SNAP 7A,C (10 Watt, Sr-90), SNAP 7E (6.5 Watt, Sr-90) and SNAP 7B,D (60 Watt, Sr-90). The performance of some of them now in operation is described with particular regard to power levels, efficiency, lifetimes, reliability and safety. Expected improvements in technology and unique applications of these devices are described.

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Abstract

REACTOR DIRECT-CONVERSION UNITS

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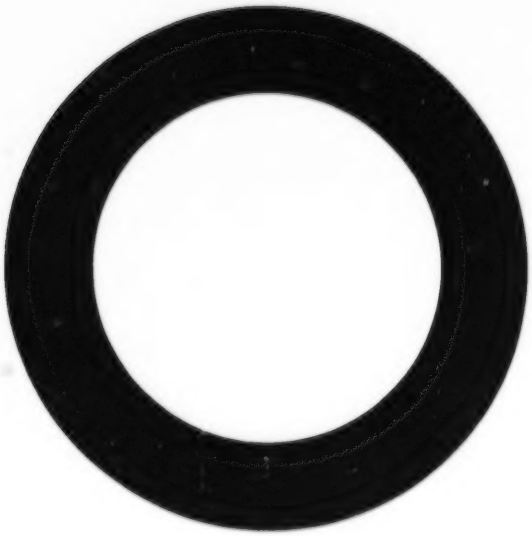
The objective of the United States Atomic Energy Commission's SNAP, Systems for Nuclear Auxiliary Power Reactor Program is to make compact, high-powered, long-lived, power sources available for the utilization of space through the application of nuclear heat sources. The program encompasses reactor powered units with both direct and dynamic power conversion in the subkilowatt to megawatt range of electrical power. The detailed objectives of the space application, i.e., launch ruggedness, high vacuum, zero gravity, radiative heat rejection, micrometeoroids, minimum weight, reliability, long life, and safety impose unique requirements on the reactor heat source and the power-conversion concept selection, design, development, flight qualification, and application integration.

A major initial objective of the SNAP program is the development and demonstration of the 500-watt thermoelectric system, SNAP 10A. The SNAP 10A system employs a 250 lb (114 kg) reactor having a homogeneous core of fully enriched U-235 and zirconium hydride moderator (H/U = 50) and a thin 2 in. (5.1 cm) Be reflector annulus which includes provisions for reflector geometrical variations for reactor control. After startup in orbit, the reactor is inherently self controlled during its lifetime by its negative temperature coefficient of reactivity. The reactor power of 35 kW(th) at an outlet temperature of 1000F (540C) is coupled to the thermo-electric converter by a NaK-78 heat-transfer loop. The pumping power (6.5 watts) is provided by a thermocouple-powered dc conduction electro-magnetic pump. The thermoelectric converter consists of 1440 couples of doped Si-Ge alloy ($Z = 0.65 \times 10^{-3}$ C-1) which are electrically connected into a series parallel circuit to provide

500 watts at 28 volts dc.

The hot junction of the thermocouples is electrically insulated from, but thermally bonded to, pipes containing the NaK-78 coolant at an average temperature of about 950F (510C). The cold junction of the thermo-couples is formed by 1440 individual aluminum radiator platelets. Heat rejection to space from the 62.5 ft.2 (5.8 m2) of integrated radiator area of these platelets maintains the cold junctions at an average temperature of 615F (325C). The complete SNAP 10A system weighs 950 lb (430 kg) which includes 215 lb (98 kg) of LiH shadow shield to reduce the radiation dose at the payload interface to less than 10^{13} nvt and 10^{8} rads per year. The system is approximately conical in shape, 5 ft. (1.5 m) in diameter by 12 ft. (3.65 m) high, with the reactor at the apex and the integral thermoelectric converter radiator located on the outer surface of the cone. The program is in the final stages of ground demonstration and pre-flight qualification.

The initial conceptual design studies for nuclear space power began in 1953. The SNAP hydride-reactor concept was chosen in 1955 on the basis of minimum weight, applicable temperature capability, and reactor and materials state of the art. The program achieved development status in early 1957 and reactor concept verification was accomplished with a critical assembly in October 1957. The initial SNAP system concept utilized a mercury Rankine cycle for power conversion. A continuing objective of the SNAP program is to develop the reactor and rotating machinery technology necessary to realize the performance advantages of mercury Rankine-cycle systems. In the program to date, two power test reactors have operated at the 50 kW(th) and 1200F (650C) requirements of a 5-kW electrical mercury Rankine system and a third reactor is operating at the 450 kW(th) and 1300F (705C) requirements of a 35 kW(e) system.



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THERMIONIC NUCLEAR REACTORS

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In the thermionic reactor concept, the heat of nuclear fission is transformed directly into electric energy by the process of thermionic emission of electric current. The system is characterized by very high power densities and operating temperatures. Required source temperatures are in the range from 1200C to 1800C. Heat rejection occurs in the range from 400C to 1000C. Applications include compact space-power units and topping units for conventional power systems. In both cases the high source and rejection temperatures indicate potential system advantages.

Two different system concepts are under study. In the out-of-pile concept, heat is extracted from a reactor heat source and is delivered to an array of thermionic converters. In the in-pile concept, the reactor fuel element essentially forms the high-temperature electrode of the thermionic converter. The in-pile design has been studied in some detail, particularly fuel-element designs and heat transmission by conventional and novel means. For the space-reactor system, the radiator, power conditioning equipment, and shielding require consideration as important contributors to system weight.

Considerable research emphasis has been placed on the development of a basic understanding of the thermionic converter. Control of the properties of the emitter and collector surface by cesium absorption has been extensively studied. Understanding of plasma dynamics is made difficult by discharge effects occurring in most ranges of efficient operation. Limitations to converter and system performance are set by high-temperature properties of materials. Screening and evaluation of a large variety of nuclear fuel and

cladding have been performed for the thermionic temperature range from 1200C to 2000C. Insulators, seals and envelope materials operable up to 1000C have been investigated.

Converter testing results for both out-of-pile and in-pile generators are summarized as a basis for establishing the present state of development of nuclear thermionics. The techniques of thermionic in-pile testing have been reviewed. Out-of-pile cell testing periods exceed several thousands of hours. In-pile testing at high performance has been accomplished for periods of several hundreds of hours, with 1000 hours of continuous testing as a current program goal.

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NUCLEAR REACTORS APPLIED TO WATER DESALTING

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In many areas of the world, available water supplies are inadequate to meet future needs. Especially critical water shortages exist in the Southwestern United States, Middle East, and North Africa.

Various alternative ways for supplementing natural water supplies have been and are under active study in the United States, chiefly by the Department of Interior, Office of Saline Water. One promising way is the production of water and by-product electricity in sea water desalting plants using reactors as the energy source.

Dual purpose plants offer more economic promise than single purpose water plants due to thermodynamic considerations and economies gained by the combination of two processes together in one larger plant. As in any dual purpose process, careful attention must be given to the method of cost allocation between the two products as the value assigned to one can have a significant effect on the cost of the other.

The Oak Ridge National Laboratory stimulated interest in the dual purpose water-power application through their studies which suggested that large nuclear reactors could produce both power and water at very low costs. These low costs resulted from the predicted sharp reduction in unit capital cost and fuel cycle cost, gained by scaling to very large sizes.

Since this initial stimulus, a number of economic and technical studies, covering a wide range of plant sizes has been conducted. A special interagency task group, involving the Atomic Energy Commission, Department of Interior, and Federal Power Commission, was organized under the auspices of the Presidential Office of Science and Technology. This group examined large

dual purpose plants using reactors from 1500 to 25 000 MW(th). Some 47 cases were analyzed with the by-product electrical outputs varying from 180 to 5950 MW(e) and water outputs from 170 to 2200 million gallons per day. The study showed that water could be produced at plant costs in the range of 23 to 33 cents per thousand gallons, assuming public financing. If more favorable financing were assumed or if the power were valued higher than in the study, water costs could be reduced to 10 to 20 cents per thousand gallons. Conveyance costs, which are quite dependent upon location and quantity of water handled, must be added to the above costs.

Of particular interest internationally is a scoping type of study being conducted for AEC and Interior for plants using 200 to 1500 MW(th) reactors. Three operational cases are being considered (water only, power only, and dual purpose) and six types of reactors are being evaluated (gas cooled, heavy water, light water, sodium graphite, and spectral shift).

The chief activity directed toward demonstrating technology has been the proposed project for the Florida Keys. An economic feasibility study has indicated that a plant producing 50 MW(e) and 10 million gallons of water per day offers an economical solution to their water shortage. Plans to proceed with construction of such a plant are currently under consideration.

Any reactor program for desalting plants must be coordinated with evaporator development and integrated into both the overall reactor and water programs for the area or country. Detailed technical and economic analysis is needed, showing comparisons with alternative methods of bringing water and power to the area, before embarking on specific projects.

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THE OAK RIDGE HIGH FLUX ISOTOPE REACTOR

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The High Flux Isotope Reactor (HFIR), now being constructed at the Oak Ridge National Laboratory, is designed primarily for the production of macro-quantities of transplutonium elements. Versatility for research is provided by inclusion of beam tubes, irradiation facilities, a variety of reflector through-holes and provision for hydraulic rabbit tubes.

The reactor is of the "flux trap" type with normal water in the central moderator-reflector trap. Because of the sensitivity of transplutonium isotope production rates to thermal-neutron flux, the design of the core was dictated by the requirement for the highest practical thermal neutron flux in the plutonium target. The average thermal-neutron flux in a 300-gram Pu-242 target is estimated to be 2 - 3 x 10^{15} n/cm^2.sec with a maximum unperturbed flux of about 5 x 10^{15} at the design power of 100 Mw(th). This high flux was obtained by optimizing the size of the island, by making the fuel region as compact as possible and the metal-to-water ratio as large as possible within limitations set by heat-removal considerations, and by reducing the ratio of maximum to average power density.

The fuel is highly enriched (93%) U-235 contained in essentially one fuel assembly consisting of two concentric annuli of involute-shape plates. The dimensions of the active portion of the cylindrical core are 5.6 in. (14.2 cm) inner diameter, 16.5 in. (41.9 cm) outer diameter and 20 in. (50.8 cm) high. The plates contain U_3O_8 in an aluminum matrix, clad with aluminum. Heat is removed by pumping about 15 000 gal(US)/min. of normal water under

600 lb/in.^{2} (42.2 kg/cm^2) pressure to produce a coolant velocity of 42 ft/sec (10.7m/sec) downward between the plates. The average and hot-spot maximum heat fluxes are 8 x 10^5 and 2 x 10^6 Btu/h.ft^2, respectively. A cylindrical beryllium reflector surrounds the fuel. With a fuel loading of 9.4 kg of U-235 the average fuel cycle time will be about 15 days.

The ratio of maximum to average power density is held to about 1.5, excluding hot-spot effects, by (1) cylindrical core geometry; (2) reflector control in the form of two concentric, oppositely driven, cylindrical control plates located in the annulus between the fuel and the beryllium reflector and containing an axially graded poison; (3) extending the non-fuel-bearing portions of the fuel plates; (4) radially graded fuel concentration in the plates; and (5) inclusion of radially graded burnable poison in the plates. An extensive investigation was made of means to increase the ratio of surface area to volume and to increase the maximum permissible heat flux. Included were analyses and measurements of fuel-plate strength, burnout behavior, aluminum-oxide formation rates and effects on heat transfer, fabrication tolerances, control-system characteristics and the elimination of side-plates in the annular fuel assembly.

Critical experiments have verified the calculated value of 5 x 10^{15}n/cm^2.sec for the maximum thermal-neutron flux at 100 MW. Based on fuel-element development and the fabrication of test cores, it is believed that the necessary close dimensional and fuel-distribution tolerances will be achieved.

The reactor will be completed and operation started during 1965. The cost of the reactor including building and auxiliary facilities will be slightly less than \$15 x 10^6.

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THE BROOKHAVEN HIGH FLUX BEAM RESEARCH REACTOR

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The High Flux Beam Reactor (HFBR) has been designed and constructed to meet current needs in neutron physics and chemistry research at Brookhaven. Sixteen experimental facilities have been provided in the reactor. Seven of these are irradiation facilities, four for fast neutron exposures, and three for thermal neutron exposures. Nine beam tubes are provided in the reactor vessel and shield for the extraction of external neutron beams. The reactor core, reflector, and beam tubes are arranged to enhance the low energy neutron flux in the beams, while decreasing the fast neutron background.

Double polarization experiments, "cold" neutron studies of liquids and solids, and capture gamma ray spectroscopy with the samples in an external beam will all benefit particularly from the large beam intensity and low background of the reactor. Other experiments planned for the initial operating period include two and three crystal diffraction studies, polarized neutron experiments with magnetic materials, and fast chopper experiments with a new three-rotor apparatus.

The HFBR is moderated, reflected, and cooled by D_2O. The fuel is fully-enriched uranium, alloyed with aluminum and silicon, and clad with a wrought magnesium-silicon-aluminum alloy. The fuel elements are of the pressure-tube type, with curved fuel plates. At the normal operating power of 40 MW the maximum fuel element heat flux is 1.6×10^6 Btu/hr-ft^2, the maximum fuel plate surface temperature is 360F, and the maximum thermal neutron flux in the region of the beam tubes is 7×10^{14} n/cm^2-sec. The coolant velocity in the fuel element water channels is 35 ft/sec. The total coolant circulation rate is 16 600 gpm, with a core inlet temperature of 120F and a system pressure 200 psig. The reactor is controlled by 16 angle-shaped neutron-absorbing blades located outside the active core. Neutron flux perturbations

at the experimental facilities are minimized by operating half of the control blades above the core and half below, in positions which are symmetrical with regard to the core center.

The reactor vessel is constructed of a magnesium-silicon-aluminum alloy to design conditions of 275 psig pressure and temperatures of 400F for the beam tubes and 250F for the main vessel wall. The beam tubes are of the same alloy as the vessel and are welded permanently in place. The vessel head is of stainless steel. The reactor is housed in a three-level steel building, hemispherical in shape with a diameter of 176 ft - 8 in. All reactor operations, as well as the handling of samples from the irradiation facilities are carried out on the upper and lower levels, leaving the external beam experimental area on the middle level free of radioactive contamination and unnecessary traffic.

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Abstract

THE NRTS ADVANCED TEST REACTOR

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The Advanced Test Reactor, located at the National Reactor Testing Station in Idaho, was designed to provide large-volume engineering test facilities at perturbed thermal-neutron flux levels greater than 10^{15} nv. Forty curved-plate aluminum/uranium fuel elements provide nine spaces each capable of accommodating an experimental loop. Properties resulting from the novel design include: 1) negligible variation in neutron flux with time, 2) small and constant spatial flux gradients, 3) spectral adjustment for each experiment, 4) independently adjustable flux levels in five of the test positions, and 5) negligible interaction between experiments.

The four-foot-long active core is arranged to provide six-inch-diameter test spaces that are nearly optimum for experiments with cylindrical symmetry. The loops may penetrate both the top and bottom reactor vessel closures as well as nozzles penetrating the side wall, thereby providing for a variety of loop designs. Control drives are disposed so as to cause a very small degree of mechanical interference with the loops.

The first complement of experiments was integrated into the reactor design using the experience gained during the many years of operation of loops in the Materials Testing Reactor and Engineering Test Reactor. Table I summarizes the principal final design characteristics of the ATR.

The shutdown control elements consist of safety rods, the hafnium sections of which are fully withdrawn from the core during operation, and hence do not introduce flux distortions. The operational control elements in the interior and outer reflector regions extend the full length of the core, thereby maintaining a symmetrical vertical flux pattern about the midplane, with an axial maximum-to-average value less than 1.4 throughout the entire seventeen-day

operating cycle. The control elements in the inner and outer regions can be varied independently to produce power shifting and hence accomplish experimental-flux control.

The reactor building provides an operating area of great experimental flexibility and convenience. The arrangement of the reactor, fuel-handling canal and experimental cubicles provides generous space allotments to both experiment and reactor operating functions. Ordinary utilities such as cooling water, electrical power, air, and hot and warm waste systems are available in sufficient capacity so that all common needs can be supplied to the loop facilities.

A Critical Facility, which is a nuclear duplicate of the ATR, is incorporated in a portion of the canal for experiment design, fuel loading, and physics studies applicable to the main reactor.

TABLE I

FINAL DESIGN CHARACTERISTICS OF THE ATR

Power level (megawatts)	250
Fuel region volume (liters)	262
Length of active core (feet)	4
Fuel loading (kilograms U-235)	35
Maximum power density (megawatts per liter)	2.8
Primary cooling water (U.S. gallons per minute)	47 000 - 53 000
Velocity in fuel element channels (feet per second)	44 - 46
Inlet temperature (degree Fahrenheit)	130
Core inlet pressure (pounds per square inch guage)	355
Core pressure drop (pounds per square inch)	100
Maximum thermal flux in experiments (perturbed)	1.5×10^{15}
Maximum fast flux (greater than 0.6 electron volts) in experiments	2.0×10^{15}

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Abstract

CRITICAL HEAT FLUX CONSIDERATIONS IN THE THERMAL AND HYDRAULIC DESIGN OF WATER COOLED NUCLEAR REACTORS

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The critical heat flux condition is one of the most important thermal and hydraulic limits in water cooled nuclear reactors. Its occurrence is associated with a rapid deterioration of the heat transfer process and is accompanied by temperature oscillations and a sharp temperature rise which can be detrimental to the integrity of fuel elements. This paper is concerned with three aspects of the critical heat flux condition in water cooled nuclear reactors. They are (1) the experimental measurement of the critical heat flux condition, (2) its analytical or empirical prediction, and (3) the application of these measurements or predictions to the design of nuclear reactors.

Recent measurements in simplified geometries such as circular, rectangular and annular are reviewed and analyzed. The test data cover a wide range of variables, i.e., pressure of 500 to 1500 psia, flow rate of 0.1 to 13 x 10^6 lb/hr-ft^2, hydraulic diameter of 0.18 to 1.5 inch, heated length of 24 to 262 inches, and various modes of heating. The major parameters and their effects upon the critical heat flux condition are identified and described. Comparison of the results reported for the various simplified geometries and reference to previously obtained data are used to single out the remaining important sources of uncertainties, namely the effects of unheated surface area, heat flux distribution, small hydraulic diameter, and special loop test characteristics.

Test results with a four, seven, nine, twelve, and nineteen rod assembly are covered. The effects of flow rate, system pressure, rod to rod spacing, uneven heating distribution, and heated length are identified and discussed. The important role of rod to rod spacing devices and their potential helpful or harmful contributions are emphasized. Comparison of the multirod data

with corresponding simplified geometry results shows that the internally heated annulus and the circular geometry act respectively as a lower and upper limit for the multirod results.

Recent empirical correlations and analytical predictions of critical heat flux conditions are next described. The correlations of Gambill, Green, Macbeth, and Tong, and the analytical predictions of Becker, Collier, Grace, Isbin, and Tippets are outlined. Major discrepancies among the various proposed relations are noted. Most correlations are found to apply only over a limited range of test geometry and conditions.

The application of the above experimental results and predictions to the design of nuclear reactors is discussed. First, other sources of uncertainties in design or operating calculations such as power, flow, and coolant enthalpy distribution are briefly reviewed. Next, the required design margins are evaluated in terms of the present state of knowledge of the critical heat flux condition and of the consequences of operation beyond it. Recent test data with a single and two-rod geometry are described. Temperature oscillation and film boiling coefficients prevailing beyond the critical heat flux condition are reported. A reactor test with a fuel rod operating beyond the critical heat flux is also described. These results show that under appropriate operating and design conditions the consequences of operation beyond the critical heat flux are predictable and not serious.